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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit - 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/2000-004-00

Gentlemen:

In accordance with 10CFR50.73(a)(2)(iv), enclosed is the subject report concerning a manual reactor trip.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

JDV/rhs

enclosure

IE22

cc: Mr. Ellis W. Merschoff
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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Arkansas Nuclear One - Unit 1	05000313	1 of 4

TITLE (4) Manual Reactor Trip Initiated In Response To The Rapid Closure Of The Main Turbine Steam Supply Valves Which Resulted From Failure Of The Electro-Hydraulic Control System

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	14	2000	2000	004	00	04	10	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)			
POWER LEVEL (10)	82	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Specify in
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	Abstract Below
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	and in Text

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard H. Scheide, Nuclear Safety and Licensing Specialist	501-858-4618

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On March 14, 2000, at approximately 0005, the reactor was manually tripped following a component failure in the main turbine electro-hydraulic control system. During plant startup, the main turbine steam supply valve position limits were set at 100 percent to allow full travel of the valves. When reactor power reached 82 percent during power escalation, the turbine operator identified an alarm on the plant computer alarm display indicating that the valve position limit was slightly greater than 100 percent. The operator attempted to clear the alarm by depressing the "lower" pushbutton on the panel. The third time the pushbutton was depressed, the valve position limit decreased and continued to rapidly decrease after the button was released. The decreasing valve position limit caused the main turbine steam supply valves to rapidly close and the main steam safety valves began lifting. The reactor was manually tripped and all control rods inserted into the core well within their expected drop times. The cause of this event was determined to be a stuck contact on the valve position limit pushbutton. The switch was replaced prior to plant startup.

A. Plant Status

At the time this event occurred, Arkansas Nuclear One Unit 1 (ANO-1) was at approximately 82 percent power proceeding to 100 percent power following a maintenance outage.

B. Event Description

On March 14, 2000, at approximately 0005, the reactor was manually tripped following a failure of the main turbine steam supply valve control system.

The main turbine steam supply valves control the flow of steam to the turbine in response to signals from the Electro-Hydraulic Control System (EHC) [TG]. The EHC system positions the steam supply valves in response to commands from the operator or from the Integrated Control System [JA]. The Operator Control Panel provides an operator interface with the EHC system. The panel contains pushbuttons which enter valve/system settings, monitor lights which indicate controller status, and meters which indicate valve positions, shaft speed, and generated megawatts. Two digital displays provide a continuous comparison of actual and desired setpoint.

During plant startup, the main turbine steam supply valve position limits on the Operator Control Panel were set to 100 percent to allow full travel of the valves. When reactor power reached 82 percent during power escalation, the turbine operator identified an alarm on the plant computer alarm display indicating that the position limit was slightly greater than 100 percent. The operator attempted to reduce the limit to 100 percent in order to clear the alarm. He depressed the "lower" pushbutton two times and on both occasions the limit was observed to decrease on the plant computer but remained in alarm at greater than 100 percent. When the pushbutton was depressed the third time, the valve position limit decreased and continued to rapidly drop after the pushbutton was released. The "raise" pushbutton was depressed but had no effect. The decreasing valve position limit caused the steam supply valves to rapidly close and the main steam safety valves began lifting. The control room supervisor ordered the operator to trip the reactor. All control rods inserted into the core well within their expected drop times and the main turbine tripped, as designed. Fourteen of the sixteen Main Steam Safety Valves (MSSVs) lifted and all but one reseated normally. PSV-2699 on the "A" steam header did not reseat until pressure was manually lowered to approximately 965 psig using the turbine bypass valves. Vital 4160 VAC busses A1 and A2 and non-vital 4160 VAC bus H2 fast transferred to the startup transformer, as designed; however, non-vital 4160 VAC bus H1 did not fast transfer as required, but did slow transfer to the startup transformer. The slow transfer resulted in the loss of reactor coolant pumps (RCPs) P-32A and P-32C. One RCP remained running in each loop providing forced circulation. The plant was stabilized in normal hot shutdown conditions.

After completion of equipment repairs, the reactor was restarted and the main generator was paralleled to the grid at 0243 on March 15, 2000.

C. Root Cause

Investigation into the cause of the EHC system failure identified that a stuck contact on the valve position limit "lower" pushbutton switch resulted in rapid closure of the steam supply valves which ultimately necessitated manually tripping the reactor.

The failure of non-vital 4160 VAC bus H1 to fast transfer to the startup transformer was the result of a degraded relay in the breaker's control circuitry which did not close a contact in the fast transfer circuit, as required.

A review of Safety Parameter Display System data verified that the blowdown of PSV-2699 was within acceptable parameters. The lift setpoint for this valve is 1050 psig. The valve reseated at approximately 962 psig which corresponds to a blowdown of 8.4 percent which is within the expected 3-9 percent value. Therefore, the valve was determined to be operable.

D. Corrective Actions

The defective pushbutton switch in the EHC system was replaced and satisfactorily tested.

The degraded relay in non-vital 4160 VAC bus H1 circuit breaker control circuit was replaced and tested satisfactorily.

E. Safety Significance

All safety related equipment operated as designed following the manual reactor trip and the plant was safely taken to hot shutdown conditions. Although non-vital 4160 VAC bus H1 failed to fast transfer, as designed, resulting in the loss of two RCPs, one pump remained operating in each loop and forced circulation was maintained. Therefore, this event was determined to be of minimal safety significance.

F. Basis for Reportability

Since a failure of the EHC system necessitated manually tripping the reactor, this event is reportable pursuant to 10CFR50.73(a)(2)(iv) as a manual actuation of the Reactor Protection System.

This event was also reported to the NRC Operations Center in accordance with 10CFR50.72(b)(2)(ii) at 0211 on March 14, 2000.

G. Additional Information

A review of ANO Licensee Event Reports submitted since 1990 identified no previous similar events in which a reactor trip was initiated as a result of a failure of the main turbine electro-hydraulic control system.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].